NON-PUBLIC?: N
ACCESSION #: 9303290279
LICENSEE EVENT REPORT (LER)

FACILITY NAME: McGuire Nuclear Station, Unit 2 PAGE: 1 OF 13

DOCKET NUMBER: 05000370

TITLE: A Unit 2 Manual Reactor Trip was Initiated as a Result of
Equipment Failure.
EVENT DATE: 02/22/93 LER #: 93-01-0 REPORT DATE: 03/24/93

OTHER FACILITIES INVOLVED: DOCKET NO: 05000370

OPERATING MODE: 1 POWER LEVEL: 100%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Terry L. Pedersen, Manager, Safety TELEPHONE: (704) 875-4487

Review Group

COMPONENT FAILURE DESCRIPTION:
CAUSE: F SYSTEM: CF COMPONENT: VP MANUFACTURER: M430
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On February 22, 1993, at 0046, Unit 2 Control Room personnel reported hearing a sound resembling the "harmonic vibrations" associated with the feedwater lines when operating at reduced load. The Senior Reactor Operator (SRO) directed the Reactor operator at the Controls (ROATC) to check the position of the Feedwater Control Valves. The ROATC discovered the Steam Generator (SG) 'C' feedwater flow was low and decreasing even though the controller output was increasing. A Turbine Generator load reduction was begun to compensate for the reduced feedwater flow. When the SG level approached the trip setpoint, the SRO instructed the ROATC to manually trip the Reactor which in turn caused an automatic Turbine trip. Subsequent investigation revealed that valve 2CF-20 (SG 'C' Feedwater Control Valve) failed to open on demand due to a signal air leak in the valve positioner pneumatic relay. Unit 2 was operating in

Mode 1 (Power Operation) at 100 percent power at the time of the event. A cause of Equipment Failure has been assigned to this event. Corrective actions include replacement of all Feedwater Control Valve positioner pneumatic relays. In addition, a group of Nuclear Generation Department personnel previously working with the Secondary Valve Reliability Team are accelerating their review of problems associated with the Feedwater Control Valves in order to identify and correct the underlying causes.

END OF ABSTRACT

EVALUATION:

Background

The Main Feedwater (CF) system EIIS:SJ! supplies feedwater at the required temperature, pressure and flow rate to the Steam Generators (SGs) EIIS:SG!, to maintain the proper water levels with respect to Reactor EIIS:RCT! power output and Turbine (EIIS:TRB! steam requirements.

Valve 2CF-20 is one of four (one per SG) air diaphragm Feedwater Control Valves EIIS:FCV!. In the automatic mode of operation, the valve is regulated by the Feedwater Control System EIIS:JB! using feedwater flow, steam generator water level, and steam flow as control parameters. In the manual mode of operation, the valve is controlled by the Control Room EIIS:NA! Operator with the manual loader provided on the control board EIIS:MCBD!.

The valve positioner on the Feedwater Control Valves is used to position the valve dependent on the requirements of the Feedwater Control System.

Moore Products, Model 72P315 valve positioners are currently in use at this facility. This valve positioner uses a metal bellows-type pneumatic relay to compare the valve position to the required position based on monitored plant parameters. If an error exists, the relay will change output to reposition the valve to the correct position.

Description of Event

On the morning of February 22, 1993, at approximately 0045, the Control Room Senior Reactor Operator (SRO) noticed a noise that sounded like the "harmonic vibration" associated with the CF feed lines when the unit is operating at a reduced load. The SRO directed the Reactor Operator (RO) to check the CF Control valve positions. The RO immediately noted that

the CF flow to SG 'C' was low and decreasing with a mismatch in feedwater flow and steam flow. The SRO instructed the RO to commence Turbine Generator EIIS:TG! load reduction at the rate of 50 megawatts per minute (Mwe/min). The load reduction was subsequently increased to 100 Mwe/min. The level in SG 'C' was approaching the SG Level Trip setpoint of 40 percent Narrow Range (NR) Level due to the flow mismatch of 90 percent steam flow and 40 percent feed flow to SG 'C'. At this point, the SRO directed the RO to manually trip the Unit 2 Reactor. The Reactor Trip Breakers were manually opened at 0046:48 and the Auxiliary Feedwater (CA) EIIS:BA! system automatically started to restore SG 'C' level.
Operations personnel entered Emergency Procedure EP/1/A/5000/01, Reactor Trip or Safety Injection. Subsequently Operations personnel

parameters, with the exception of SG 'C' NR level, were stabilized at no load conditions within 30 minutes following the trip. SG 'C' NR level (no load conditions) was achieved within 1 hour with no significant problems. The unit was stabilized in Mode 3 (Hot Standby). The required 4 hour Notification to the NRC was made at 0155 in accordance with Procedure RP/O/A/5700/10, NRC Immediate Notification Requirements.

Operations and Instrumentation and Electrical (IAE) personnel began an investigation to determine the cause of the event. It was determined that valve 2CF-20, Feedwater Control Valve, had failed closed, thus decreasing feedwater flow to SG 'C'. Emergency Work Order number 93014676 was written to investigate/repair valve 2CF-20. Fuses supplying power to 'A' and 'B' Train solenoid valves were checked and found to be good. Due to the number of previous problems associated with Feedwater Control Valves, the IAE technicians normally assigned to the CF system were called in to perform a more thorough investigation. It was initially suspected that the pneumatic relay (Moore Products Model 72P315) on the valve positioner for valve 2CF-20 had failed. The IAE technicians simulated an open signal on the valve positioner and monitored valve position. Even with maximum pressure applied to the pneumatic relay the valve would only open approximately 25 percent. A continuous sound of air leaking from the pneumatic relay could be heard. The pneumatic relay on valve 2CF-20 positioner was replaced. A white powder was discovered inside the instrument air ports on the pneumatic relay and in the air line fittings. The IAE technicians removed the filter from the filter regulator upstream of the positioner. There was no sign of the white powder in the filter or the instrument tubing. As a precautionary measure, the pneumatic relays on the other three Unit 2 CF Control Valves were replaced. The four pneumatic relays were examined by Component Engineer 'A'. The examination of the other three pneumatic relays showed no signs of the substance found in the pneumatic relay on valve 2CF-20. The pneumatic relay from valve 2CF-20 was sent to Duke Power's Analytical & Predictive Technologies - Metallurgical Lab for analysis. The Reactor returned to Mode 1, Power Operation, on February 22, 1993 at 1902.

Conclusion

A cause of Equipment Failure has been assigned to this event because the control air sent to the valve positioner was escaping through a crack in the bellows EIIS:BE! of the pneumatic relay. Although Equipment Failure was the cause of this event, the high vibration in the valve and associated piping system are a concern. The Metallurgical Lab personnel found evidence of stress Corrosion Cracking (SCC) (see Metallurgical Analysis Report attached). SCC requires stress, a corrosive anion, and an electrolyte. Stress was provided by both residual and operational stresses. The corrosive anions were most likely provided by excess solder flux (white powder) which was found throughout the bellows. The flux residue was determined to contain Chloride and was acidic (pH of approx. 1). Although Chloride is corrosive to brass, it is not a typical actor in its stress corrosion. Anions

cause SCC in brass (e. g. ammonia) may not have been detected by the methods available. The electrolyte was provided by moisture which probably got into the control instrument air line periodically, as evidenced by the splotchy blue stains on the exterior of the inner bellows of the pneumatic relay.

Prior to this event there had been at least one other event (LER 370/92-04) associated with the Feedwater Control Valves involving a failed valve positioner. The apparent root cause of that event was Equipment Failure due to possible installation deficiency. The clevises on the ends of the valve positioner linkage were not in the same plane when assembled. Movement of the positioner arm and linkage then resulted in excessive wear and caused failure of the clevis. In both events, conditions existed that alone may not have caused problems. However, excess movement/vibration contributed to problems that resulted in failure. Continued high vibration could lead to other types of failures. A further detailed review of all problems associated with the Feedwater Control Valves, will be performed by a combination of personnel from Operations, Engineering and Maintenance.

As a result of this event, Component Engineering personnel were considering the possibility of replacing the metal bellows-type pneumatic relay (Moore Products Model 72P315) with one containing a viton elastomer diaphragm (Moore Products Model 721P315) during the next scheduled refueling outage (2EOC8). However, during this investigation another pneumatic relay failure on valve 2CF-23 (SG 'B' Feedwater Control Valve) resulted in another unit trip. While the unit was off line, the metallic bellows-type pneumatic relay was replaced on all four CF Control Valves with the pneumatic relay utilizing the elastomer diaphragm.

A review of the operating Experience Program (OEP) Database for the twenty-four months prior to this event revealed one event involving a Reactor trip in which the cause was a failed valve positioner. This event, documented on LER 370/92-04, coupled with the subsequent event indicate a recurring problem. The corrective actions for the previous event were specific to the failed component and would not have prevented this event from occurring.

This event is Nuclear Plant Reliability Data System (NPRDS) reportable, due to the failure of valve 2CF-20 maintaining it's desired position.

There were no personnel injuries, radiation overexposures, or uncontrolled releases of radioactive material resulting from this event.

CORRECTIVE ACTIONS:

Immediate: 1) Operations personnel implemented the Reactor Trip or Safety Injection procedure.

Subsequent: 1) Work Control personnel initiated emergency Work Order number 93014676 to investigate/repair valve 2CF-20.

2) IAE technicians replaced the metal bellows-type pneumatic relay on all four Unit 2 CF Control Valves.

3) Personnel working with the Secondary Valve Reliability Team are accelerating their review of past performance problems associated with the Feedwater Control Valves to identify probable causes.

Planned: 1) Systems Engineering personnel will coordinate the replacement of the metal bellows-type pneumatic relays with Viton elastomer diaphragms.

2) Component Engineering personnel will implement approved actions to correct the causes identified by the team

referenced in Subsequent Action 3 above.

SAFETY ANALYSIS:

During this event, the valve positioner associated with valve 2CF-20 failed to maintain its desired position causing a decrease in SG 'C' feedwater flow. This in turn caused a decrease in SG 'C' level, requiring valve 2CF-20 to open. Valve 2CF-20 did not open on demand and the control circuitry continued to send an "open" signal to the positioner in an effort to maintain feedwater flow. The valve failed to respond properly because the control instrument air sent to the valve positioner was escaping through the crack in the bellows of the pneumatic relay. The control circuitry performed as designed. As the level in SG 'C' approached the 40 percent setpoint, the Reactor Trip breakers were manually exercised in order to avert a challenge to the plant safety systems. This event is bounded by the Turbine Trip Accident found in Chapter 15, subparagraph 15.2.7 of the McGuire Final Safety Analysis Report.

Primary and Secondary system no load conditions necessary to achieve a safe shutdown were attained within 30 minutes of the manual Reactor trip with the exception of SG 'C' NR level. The SG 'C' NR level no load condition was achieved approximately 1 hour after the trip. This event presented no hazard to the integrity of the plant.

The health and safety of the public and plant personnel were not affected as a result of this event.

ADDITIONAL INFORMATION

Sequence of Events:
PTR - Post Trip Report
SSL - Unit 2 Shift Supervisors Logbook
ROL - Reactor Operator Logbook
WO - Work Order
ER - Unit 2 Events Recorder
PR - Personnel Recollection
PRO - Procedure
OAC - Operator Aid Computer

Date Time Event

2/22/93 0045 Flow to 'C' Steam Generator was discovered decreasing rapidly. Turbine load reduction was initiated at 50 Mwe/min and increased to 100 Mwe/min. (SSL, ROL)

0046:48 The RO manually tripped the reactor. (SSL, ROL, ER)

0046:53 The Auxiliary Feedwater Pumps 'A' & 'B' automatically started. (OAC)

0155 The NRC was notified as required by RP/O/A/5700/10. (PRO)

1230 Change out of the metallic bellows-type pneumatic relay on the CF Control Valves was completed. (PR)

1902 Unit 2 Reactor returned to Mode 1. (ROL, SSL)

METALLURGY
DUKE POWER
FAILURE ANALYSIS

APPLIED SCIENCE CENTER

Metallurgical Analysis Report

Sample No.: 1436 Station: McGuire Unit: 2
Requestor/Dept.: Jeff Miller - MNS
Principal Investigator: Sue Anderson
Submitted To: Jeff Miller Date: 3/4/93
cc: Donna Keck - NGD Nuclear Services
Jim Algood - MNS Safety Review

Equipment Description:
Bellows-type relay from valve positioner on 2CF20

Background Information:
The "C" steam generator feedwater regulation valve 2CF20 failed closed, leading to a manual trip. The bellows-type relay in the Moore valve positioner was determined to have failed. Due to previous problems with relays of this type, each had been replaced in kind during the previous outage. A failure analysis was requested.

In addition, an unidentified white powder was found in the port for the control air signal line. EDS qualitative analysis of this material was requested.

Description/Macro-Examination:

The material in the control air line only partially occluded the port, as viewed from the manifold block (Figure 1). The material was a fine yellowish-white powder and was not magnetic.

The outer shield of the relay was sectioned off through the solder joint. The outer bellows, which was generally clean, was sectioned through the base solder joint and around the cap at top (Figure 2). The inner bellows was cracked below the second convolution from the bottom (base plate) side, in the transition from the flat convolution underside to the radius (Figure 3). The crack covered approx. 100 degrees of the circumference. While the metal was generally shiny, several dark blue stains appeared around the crack, one in the center of its length (Figure 4).

The inside of the outer bellows and the outside of the inner bellows were dusted with the same yellowish-white powder as was found in the control air port. The base plate was caked with the powder in some areas and had several patches of orange-brown resinous solder flux visible (Figure 5). It appeared that the powder was original solder flux, possibly degraded or dried, that had not melted to form the orange-brown resin.

Fractography:
The crack was pulled apart to examine the fracture surface. Scanning electron microscope examination located many regions of distinct fatigue striations along the crack length (Figure 6).

Metallography:
A cross-section was taken through the center of the crack near the blue stain. Two branching cracks were found extending from the exterior surface of the inner bellows, parallel to the main crack (Figure 7). The small cracks were characteristic of stress corrosion cracks; once one grew to a critical flaw size, its mode of growth changed to fatigue and propagated through-wall.

Chemistry/Mechanical Testing:
Qualitative chemical analysis using EDS was performed on the yellowish-white powder using two methods. Both indicated that the material was resinous or organic (carbon-rich) and contained some amount of chloride (Figure 8).

Chloride is not typically an actor in the stress corrosion of brass. Ammonia compounds are known to cause SCC in brass, but ammonia is undetectable by EDS. Heating and precipitation/wet chemical testing was used to qualitatively indicate whether ammonia compounds were present in the resinous flux residue. No ammonia was detected; the flux residue was determined to be fairly acidic (pH of approx. 1).

Conclusions:
The crack in the inner bellows of the valve positioner relay appeared to have been initiated by stress-corrosion cracking, as evidenced by the presence of branching cracks near the center of the fracture. One crack which reached critical flaw size then propagated through fatigue under vibrational stresses, as shown by the presence of fatigue striations elsewhere on the fracture surface.

Stress-corrosion cracking of a metal requires stress, a corrosive anion, and an electrolyte. The stresses in the bellows were likely both residual and operational. The soldering flux was a source of corrosive anions; while chloride is corrosive to brass, it is not a typical actor in its stress corrosion. Anions known to cause SCC in brass (e.g. ammonia) may not have been detected by the methods available. The splotchy blue stains indicated that moisture was periodically present on the exterior of the inner bellows, providing the electrolyte.

It should be considered that it would be possible to leave the flux residue in the relay and avoid developing a stress corrosion initiator. This condition would require that the flux not be reactivated with an electrolyte. The instrument air would have to be free of moisture to avoid this condition.

The white powder in the control air port and around the inner bellows appeared to be related to the flux used in soldering the bellows. It was believed to be flux which had not liquified during soldering; the melted flux residue was orange-brown and resinous. It was found to contain chloride.

The evidence supports the root cause of fracture to be a combination of a manufactured condition (solder flux residue) and an operational condition (moisture in the air line). Stress corrosion cracks grew in the presence of the flux residue and moisture, with one then propagating as a fatigue crack and failing the bellows. These conditions could exist in replacement parts of the same design and manufacture.

It was reported that vibrational stresses monitored in the valve reached values in the range of 9G during certain periods of operation. Stresses of this magnitude could be sufficient to initiate fatigue cracks at the geometrical stress concentration of the convolution. It is possible that fatigue cracks could initiate in a bellows of this type in the absence of stress corrosion cracks. In this particular case, the presence of multiple branching cracks around the crack origin indicated that fatigue cracking was not responsible for crack initiation but was the mode of crack propagation.

It has been proposed to replace the metal bellows-type relay with one containing a viton elastomer diaphragm. The diaphragm would not be as sensitive to vibrational fatigue as the bellows and should offer better corrosion resistance.

If the Metallurgy Lab can be of further assistance, please call me at (704) 875-5326.

Approved by: Date:

Reviewed by:

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Page 3, Sample #1436

Figure 1 "View of manifold block of valve positioner, showing extent of yellowish-white material in control air port. Ma-2497." omitted.

Figure 2 "Overall view of failed valve positioner relay, as received. Shield and outer bellows were cut away as shown to reveal inner bellows. Ma-2496" omitted.

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Figure 3 "Inner bellows revealed by sectioning. Crack is visible under second convolution from bottom (arrows). Gauge = 1/16". Ma-2500" omitted.

Figure 4 "Detail of crack from Figure 3. Metal is mostly shiny but a stain is visible near the center of the crack." omitted.

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Figure 5 "Base plate with inner bellows sectioned off. ARea 1 shows flux partially converted to white power; Area 2 shows flux mixed with solder. Gauge = 1/16". Ma-2501." omitted.

Figure 6 "Fatigue striations visible on fracture surface of crack in Figure 4, broken open. Crack traveled from bottom to top. (1800X) S-1888." omitted.

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Figure 7 "Cross-section through inner bellows near center of crack. Two branching cracks extend from OD, parallel to fracture surface (right). (500X) Mi-2632." omitted.

Figure 8 "EDS spectrum produced by yellowish-white powder, mounted to stub with carbon paste. Carbon is the major signal but chloride is distinctly present." omitted.

Duke Power Company (704)875-4000
McGuire Nuclear Station
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

DUKE POWER

March 24, 1993

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: McGuire Nuclear Station Unit 2
Docket No. 50-370
Licensee Event Report 370/93-01

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 370/93-01 concerning a Unit 2 Manual Reactor Trip. This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

T.C. McMeekin

TLP/bcb

Attachment

xc: Mr. S.D. Ebneter INPO Records Center
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U.S. Nuclear Regulatory Commission NRC Resident Inspector
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